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10 CFR 50.73

October 12, 2020
Serial: RA-20-0275

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400/Renewed License No. NPF-63

Subject: Licensee Event Report 2020-002-00

Ladies and Gentlemen:

Duke Energy Progress, LLC, submits the enclosed Licensee Event Report 2020-002-00 in accordance with 10 CFR 50.73 for Shearon Harris Nuclear Power Plant, Unit 1 (HNP). On August 13, 2020, a control rod dropped during control rod drive system surveillance testing, which required a manual reactor trip in accordance with plant procedural guidance. All safety systems functioned as expected. The cause of the control rod drop has been addressed and the unit has been restarted. This event had no significance with respect to the health and safety of the public.

There are no regulatory commitments contained within this report.

Please refer any questions regarding this submittal to Sarah McDaniel at (984) 229-2002.

Sincerely,

Kim E. Maza

Enclosure: Licensee Event Report 2020-002-00

cc: J. Zeiler, NRC Senior Resident Inspector, HNP
M. Mahoney, NRC Project Manager, HNP
NRC Regional Administrator, Region II

NRC FORM 366 (08-2020) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See Page 3 for required number of digits/characters for each block)
(See NUREG-1022, R.3 for instruction and guidance for completing this form
http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2023

1. Facility Name	2. Docket Number	3. Page
Shearon Harris Nuclear Power Plant, Unit 1	05000 400	1 OF 3

4. Title
Manual Reactor Trip Due to Control Rod Drop During Surveillance Testing

5. Event Date			6. LER Number			7. Report Date			8. Other Facilities Involved	
Month	Day	Year	Year	Sequential Number	Revision No.	Month	Day	Year	Facility Name	Docket Number
08	13	2020	2020	- 002 -	00	10	12	2020		05000
									Facility Name	Docket Number
										05000

9. Operating Mode	10. Power Level
1	100

11. This Report is Submitted Pursuant to the Requirements of 10 CFR §: *(Check all that apply)*

10 CFR Part 20	☐ 20.2203(a)(2)(vi)	☐ 50.36(c)(2)	☑ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(A)	**10 CFR Part 73**
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.69(g)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(4)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(i)	**10 CFR Part 21**	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(1)(i)
☐ 20.2203(a)(2)(ii)	☐ 21.2(c)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(i)
☐ 20.2203(a)(2)(iii)	**10 CFR Part 50**	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)	☐ 73.77(a)(2)(ii)
☐ 20.2203(a)(2)(iv)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)	
☐ 20.2203(a)(2)(v)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)	

☐ **OTHER** (Specify here, in abstract, or NRC 366A).

12. Licensee Contact for this LER

Licensee Contact	Phone Number (Include area code)
Sarah McDaniel, Regulatory Affairs Engineer	(984) 229-2002

13. Complete One Line for each Component Failure Described in this Report

Cause	System	Component	Manufacturer	Reportable to IRIS		Cause	System	Component	Manufacturer	Reportable to IRIS
X	AA	75	F183	Y						

14. Supplemental Report Expected		15. Expected Submission Date	Month	Day	Year
☑ No	☐ Yes (If yes, complete 15. Expected Submission Date)				

16. Abstract (Limit to 1560 spaces, i.e., approximately 15 single-spaced typewritten lines)

On August 13, 2020 at 09:38 Eastern Daylight Time, with Shearon Harris Nuclear Power Plant, Unit 1 (HNP), in Mode 1 at 100 percent power, a control rod dropped during control rod drive (CRD) system surveillance testing. This required a manual reactor trip in accordance with plant procedural guidance. The reactor protection system (RPS) and auxiliary feedwater system (AFWS) actuated as designed. Based upon troubleshooting, it was determined that the control rod drop was due to the accumulation of crud in the dashpot region of the control rod drive mechanism (CRDM) movable gripper latch support assembly, which prevented the stationary gripper from latching onto the CRD shaft when the movable gripper was released. Three full-length exercises for each control rod bank were completed to remove crud from the CRDM surfaces that restored the CRDM to operable for unit startup. The dropped control rod was recovered from the reactor core and restored to the fully withdrawn position. A preventative maintenance activity to exercise the control rod banks 10 steps on a weekly frequency was implemented. Coil current traces will be obtained periodically during rod exercises to monitor for crud presence. After review, the CRD system surveillance test procedure was revised to remove the instruction to trip the reactor if a dropped rod is present with any control rods inserted into the core other than the controlling bank. In the next refueling outage, full length control rod exercises and flushing of applicable CRDMs will be completed. Due to actuation of the RPS and AFWS, this event is reportable per 10 CFR 50.73(a)(2)(iv).

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

(See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Shearon Harris Nuclear Power Plant, Unit 1	05000- 400	2020	- 002	- 00

NARRATIVE

Note: Energy Industry Identification System (EIIS) codes are identified in the text within brackets [].

A. Background

Prior to the event, Shearon Harris Nuclear Power Plant, Unit 1 (HNP), was operating in Mode 1 at approximately 100 percent power. There were no structures, systems, or components that were inoperable at the time of this event that contributed to the event. This event is reportable per 10 CFR 50.73(a)(2)(iv)(A) as "Any event or condition that results in valid actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of [10 CFR 50.73]…" due to actuation of the reactor protection system (RPS) [JC] and auxiliary feedwater system (AFWS) [BA]. All actuated safety systems functioned as designed.

NRC Event Notification 54834 reported this event as an unanalyzed condition that significantly degrades plant safety. Following this event, it was determined that the HNP Updated Final Safety Analysis Report, Chapter 15 dropped rod [ROD] transient analysis bounds the condition where a control rod is dropped during performance of the rod movement test used to demonstrate control rod operability. Therefore, this event is not reportable under 10 CFR 50.73(a)(2)(ii) as an unanalyzed condition that significantly degrades plant safety.

B. Event Description

On August 13, 2020 at 09:38 Eastern Daylight Time, with Shearon Harris Nuclear Power Plant, Unit 1 (HNP), in Mode 1 at 100 percent power, the rod cluster control assembly (RCCA) F2 dropped during control rod drive (CRD) system [AA] surveillance testing. The RCCA drop occurred when the operator commanded that the RCCA be withdrawn from the 225-step position. The surveillance test procedure contained guidance that stated, "During performance of this test, a dropped rod with any control rods inserted into the core other than the controlling bank is an unanalyzed plant condition and the reactor [RCT] must be tripped." In this case, control bank A was not at its park position when the RCCA F2 dropped. The reactor was manually tripped and all safety systems functioned as expected. Operations stabilized the plant in Mode 3. The RPS and AFWS actuated as designed.

Based upon post-trip troubleshooting, it was determined that the RCCA F2 drop was due to the accumulation of crud in the dashpot region of the control rod drive mechanism (CRDM) [75] F2 movable gripper latch support assembly, which prevented the stationary gripper from latching onto the CRD shaft [DRIV] when the movable gripper was released. The incomplete lift of the rod by the lift coil resulted in failure of the stationary gripper to adequately engage the lobe on the control rod drive shaft. With both the movable and stationary grippers not engaged on the CRD shaft, the control rod dropped into its fuel assembly.

This event occurred approximately 9 months after new CRDMs were placed in service as part of the HNP reactor vessel head (RVH) [RPV] replacement project. The new CRDMs were manufactured by Framatome, Model L-106A1. The accumulation of crud on newly installed CRDMs after new CRDMs are placed in service is a common condition, based upon industry operating experience. Excessive crud accumulation can inhibit proper movement of the CRDM latch assemblies, resulting in rod mis-steps and rod drops. Over time, a tightly adhered oxide layer or passivation layer develops on the CRDM latch assemblies. The eventual stable passivation layer on the latch assembly components prevents accumulation of additional crud, which will allow for proper continued CRDM operation. With the new CRDM placed in service for approximately 9 months, a passivation layer on the internal CRDM latching components had not yet been achieved, which would have prevented the accumulation of crud that caused the CRDM to malfunction. The RVH replacement project implemented original equipment manufacturer recommendations and written operational experience recommendations to reduce corrosion in CRDMs that minimize the potential of rod drops. Based upon the Westinghouse

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guidance in InfoGram 05-4, Westinghouse plants have not experienced untrippable control rods due to crud accumulation.

C. Causal Factors

The cause of the CRDM F2 malfunction was crud accumulation on CRDM internal components, which resulted in the lower stationary gripper not engaging on the CRD shaft when the RCCA was being withdrawn from the fuel assembly. Informal (unwritten) operational experience obtained from CRDM original and current manufacturers following the event indicates that periodic cycling of control rods following a RVH replacement return to service will minimize crud build-up until a passivation layer is established and crud may be monitored through coil current traces during control rod cycling. Following this event, the surveillance test procedural guidance to trip the reactor if a dropped rod is present with any control rods inserted into the core other than the controlling bank was determined to be overly conservative since this condition is analyzed by the HNP safety analysis. This guidance was added to the procedure in 2008.

D. Corrective Actions

The Westinghouse rod control diagnostic flow charts were implemented to support the causal analysis of the control rod drop. Three full-length exercises for each control rod bank were completed that restored the CRDM to operable for unit startup. Coil current traces were obtained during these exercises, which were later determined by the CRDM original equipment manufacturer to indicate the presence of crud on the CRDMs. The RCCA F2 was recovered from the reactor core and restored to the fully withdrawn position.

A preventative maintenance activity to exercise the control rods 10 steps on a weekly frequency was implemented upon reactor startup. Coil current traces will also be obtained periodically during rod exercises to monitor for crud presence. The CRD system surveillance test procedure was revised to remove the instruction to trip the reactor if a dropped rod is present with any control rods inserted into the core other than the controlling bank.

Coil current trace data from control rod exercises will be evaluated to determine if any changes to the rod exercising frequency or scope are warranted. During the next refueling outage, full length control rod exercises and flushing of applicable CRDMs through rod drops will be completed to remove crud. Coil current trace data will also be obtained. The results of the outage CRDM exercising and flushing activities will be evaluated to determine if a change in the preventative maintenance strategy is needed.

E. Safety Analysis

The primary function of the CRDM is to insert or withdraw RCCAs within the core to control average core temperature and to shutdown the reactor. The trippability of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. At no point during this event was the trippability of the shutdown rods or control rods questionable. The dropped control rod fell normally to the bottom. There was no loss in function of safety systems, structures, or components. There were no significant equipment abnormalities. The condition did not result in a safety system functional failure and had no adverse impact on the health and safety of the public.

F. Additional Information

There have been no events similar to the event documented in this LER in the past three years.